

NON-PUBLIC?: N  
ACCESSION #: 9001310028  
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Peach Bottom Atomic Power Station - Unit 2 PAGE: 1 OF 04

DOCKET NUMBER: 05000277

TITLE: Unit 2 Scram Due To Personnel Error During Testing Activities  
EVENT DATE: 12/20/89 LER #: 89-033-00 REPORT DATE: 01/19/90

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR  
SECTION:  
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:  
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COMPONENT FAILURE DESCRIPTION:  
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:  
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

#### ABSTRACT:

On 12/20/89 at 1752 hours with Unit 2 operating at 100% power, a full  
scram signal was received when a technician performing a surveillance on  
Average Power Range Monitor (APRM) "D" inadvertently operated a switch on  
APRM "A" (APRM "D" actuates a "B" channel half scram signal while APRM  
"A" actuates an "A" channel half scram signal). The reactor feedwater  
pumps (RFP) tripped following the scram. The High Pressure Coolant  
Injection (HPCI) and the Reactor Core Isolation Cooling (RCIC) systems  
actuated as designed to maintain reactor water level, other safety  
systems operated as designed. Control Rod 38-39 settled to position 02  
shortly following the scram and was later re-inserted. The root cause of  
the event has been attributed to procedural deficiencies and inattention  
to detail by the technician performing the surveillance. The technician  
involved in this event was counseled and disciplined. APRM Surveillance  
procedures which test APRM channels adjacent to other APRM channels have  
been revised to provide physical barriers between APRM channels during

testing and to instruct operators to bypass adjacent APRM channels when permissible. Current panel labeling will be evaluated and improved as appropriate. There was one previous similar event.

END OF ABSTRACT

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#### Requirements for the Report

This report is required pursuant to 10 CFR 50.73(d)(2)(iv) because an inadvertent equipment manipulation resulted in an engineered safeguard feature actuation (i.e., Reactor Protection System (EIS:SC)).

#### Unit Status at Time of Event

Unit 2 was operating at 100% power. A surveillance was in progress on Average Power Range Monitor (APRM) "D" (EIS:MON).

On 12/20/89 at 1730 hours with Unit 2 operating at 100% power, a surveillance was being performed on Average Power Range Monitor (APRM) "D". At 17:52:54 hours, with a "B" Reactor Protection System (RPS) channel half scram signal inserted per the ongoing surveillance, an "A" RPS channel half scram signal was received when one of the technicians (non-utility, non-licensed) performing the surveillance inadvertently operated a switch on APRM "A" resulting in a full reactor scram.

Five seconds following the scram signal, indicated reactor level dropped to 0 inches (approximately 172 inches above top of active fuel) resulting in a Primary Containment Isolation System (PCIS) (EIS:JM) Group II and III isolation and a Standby Gas Treatment (SBGT) System (EIS:BH) initiation. Ten seconds following the scram (17:53:04 hours) the High Pressure Coolant Injection (HPCI) System (EIS:BJ), the Reactor Core Isolation Cooling (RCIC) System (EIS:BN), and the Alternate Rod Injection (ARI) System auto initiated as designed when indicated reactor water level dropped to less than -48 inches.

At 17:53:07 hours, the A, B and C Reactor Feed Pumps (RFP) (EIS:SJ) unexpectedly tripped. At 17:53:34, reactor level returned to an indicated level of 0" and the HPCI and RCIC Systems were manually shutdown.

At 1801 hours, Control Rod (EIS:ROD) 38-39 was observed to have settled to position 02 (position 48 is fully withdrawn). At 18:15 hours reactor water level was stabilized to an indicated +24 inches, the PCIS Group II and III Isolation was reset, and the SBGT System was removed from

service. At 1825 hours, the RPS and ARI signals were reset. On 12/21/89 at 0250 hours, Control Rod 38-39 was inserted from position 02 to 00.

#### Cause of the Event

The proximate cause of the reactor scram was the inadvertent operation of the mode switch on APRM "A" while APRM "D" was in test. The root cause of the inadvertent operation of the mode switch on APRM "A" appears to be a combination of the following two conditions:

a. Inattention to detail by the technician performing the surveillance in that he did not realize that he was operating the wrong switch, although he had previously operated the correct switch during the surveillance.

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b. Procedural deficiencies in that the procedure should have specified that the adjacent APRM channel be bypassed prior to performing the test.

The cause of the RFP trip appears to be associated with the low suction pressure trip circuitry on each pump. The current setpoint for the low suction pressure trip is 300 psig with (at the time of the event) a common time delay of seven seconds for each pump. It is believed that the sudden increase in Feedwater flow Caused suction pressure to drop below 300 psig for a period of time longer than seven seconds.

The settling of Control Rod 38-39 to position 02 appears to have been caused by a reverse flow condition above the drive piston on the Control Rod Drive Mechanism (EIIS:AA). This reverse flow condition is caused by broken or badly worn stop piston seals (EIIS:SEAL). Post scram testing of Control Rod 38-39 yielded results which support this theory.

#### Analysis of Event

No actual Safety consequences occurred as a result of this event.

The Reactor Protection System operated properly throughout this event. Other safety systems operated as designed.

The unexpected trip of the Reactor FeedWater Pumps did not Cause a significant challenge to plant Safety. HPCI and RCIC actuated prior to the RFP trips and were able to maintain reactor water level as designed. Operations personnel were able to re-establish feedwater flow shortly after the scram.

The Technical Specification required shutdown margin was maintained even though Control Rod 38-39 settled to position 02. The maximum subcritical banked withdrawal position for Unit 2 for the current fuel cycle (cycle 8) is notch position 02 (185 Control Rods can be at position 02 and the required shutdown margin maintained).

General Electric Company technical information concerning Control Rod settling (Services Information Letter 52) indicates that in other than severe cases, Control Rods will settle to between positions 00-02 following a scram when broken or worn stop piston seals exist. Since post scram testing of Control Rod 38-39 yielded scram times within Technical Specification limits, it is felt that this Control Rod will not drift out further than position 02 if another scram occurs.

#### Corrective Actions

The following corrective actions have been taken:

1. The technician involved in this event was counselled on the importance of attention to detail when operating sensitive equipment and disciplined as deemed appropriate by his supervision.
2. APRM surveillance procedures (including SI-2N-60A-APRM-DICW) with similar testing conditions have been revised to add cautions about the consequences of operating the wrong switch, to provide instructions to physically cover the switch on the

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adjacent APRM channel, and to instruct the unit operator to bypass the other APRM channel on the panel in test when permissible.

3. The time delays on the low suction pressure trips on the reactor feed pumps have been changed from a common time of seven seconds to seven, twelve and fifteen seconds for the A, B, and C RFP's respectively.

The following corrective actions will be taken:

1. A team of technicians and engineers will review this event and provide recommendations for permanent labeling improvements on the APRM panels, permanent or semi-permanent physical barriers to prevent inappropriate switch operation, and changes to the surveillance procedure to minimize the length of time a half-scram signal is present because of the test.

2. The current RFP low suction pressure trip point of 300 psig will be evaluated to determine if it should be lowered to reduce the possibility of pump trips during transient conditions.

3. Control Rod 38-39 normal testing will be monitored during the current operating cycle to identify any further degradation in the Rod's performance.

4. During the next refueling outage on Unit 2, the Control Rod Drive Mechanism for Rod 38-39 will be repaired as required.

#### Previous Similar Events

One previous similar event reported in LER 2-88-02 was identified. This event involved the operation of the wrong control switch by a unit operator which resulted in a full scram signal while the unit was shutdown. The corrective actions taken as a result of LER 2-88-02 involved the stressing of attention to detail by operations personnel only and, therefore, would not have prevented this event (the technician involved in this event is part of the Maintenance/Instruments and Controls group at the site.)

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PEACH BOTTOM ATOMIC POWER STATION  
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D. M. Smith  
Vice President

January 19, 1990

Docket No. 50-277

Document Control Desk  
U. S. Nuclear Regulatory Commission  
Washington, DC 20555

SUBJECT: Licensee Event Report  
Peach Bottom Atomic Power Station - Unit 2

This LER concerns a Unit 2 Scram due to personnel error during

testing activities.

Reference: Docket No. 50-277

Report Number: 2-89-033

Revision Number: 00

Event Date: 12/20/89

Report Date: 1/19/90

Facility: Peach Bottom Atomic Power Station

RD 1, Box 208, Delta, PA 17314

This LER is being submitted pursuant to the requirements of 10 CFR 50.73(a)(2)(iv).

Sincerely,

cc: J. J. Lyash, USNRC Senior Resident Inspector

W. T. Russell, USNRC, Region I

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